M&C 2017 - International Conference on Mathematics & Computational Methods Applied to Nuclear Science & Engineering, Jeju, Korea, April 16-20, 2017, on USB (2017)

Sensitivity studies on the production of Cm isotopes in spent fuel for safeguards applications

A. Borella, A. Calleja, L. Fiorito

SCK•CEN: Boeretang 200, Mol, B2400, aborella@sckcen.be

Abstract - *The verification of spent fuel elements is a major task for nuclear safeguards inspectors. The so-called Fork detector is often used in the verification of spent fuel elements. With this device one measures the total neutron emission that is originating mainly from the spontaneous fission decay due to $^{242,244}Cm$ produced during irradiation. The neutron emission of a spent fuel element can be estimated from operator declared data by using depletion and evolution codes; the information obtained with these codes can be used together with experimental data to draw safeguards conclusions. To understand and compare the performances of measurements methods, it is important not only to have an estimation of the associated neutron emission but also of its uncertainty, which is often not given by depletion and evolution codes. The work presented here focuses on sensitivity studies to identify the nuclear data that mostly affect the production of Cm isotopes. In addition, this study assesses the impact of the uncertainty of nuclear data on the uncertainty of Cm abundances for a given irradiation case.*

The obtained results show that the contribution of the uncertainty in the decay data is smaller than the contribution of the uncertainty in the radiative capture cross section data. Uncertainties in the radiative capture cross section of ^{242m}Am and ^{241}Am are the largest contributors to the uncertainties in the abundance of ^{242}Cm, while uncertainty in the radiative capture cross section of ^{243}Am and ^{242}Pu are the main contributors to uncertainty in ^{244}Cm abundance.

I. INTRODUCTION

One of the main tasks within safeguards activities carried out by IAEA is the verification of spent fuel elements (SFE). Due to the residual fissile content, spent fuel verification is carried out to verify SFE integrity and to ensure there has been no gross diversion of material from SFE [1]. This is done by verification of operator declared values on burnup (BU), initial enrichment and cooling time (CT) after discharge [2]. The verification of operator declared values can be done by means of non-destructive analysis (NDA) techniques [1]. One of the traditional methods for the verification of the SFE is the so-called Fork detector. It relies on neutron and gamma measurements [2,3,4] and enables inspectors to verify the consistency of operator declared values on burnup and cooling time after discharge. In recent years, efforts to reproduce by means of simulations the absolute Fork detector response were carried out [3, 4]. These works rely on simulations to predict the detector response in view of understanding the observed deviations between calculated and measured observables. In this framework, an accurate determination of the composition of the spent fuel as well as an assessment of its associated uncertainty are important.

One of the observables of interest is the neutron count rate of the Fork detector. This quantity depends on the neutron emission of a SFE which is mainly due to Cm even isotopes [5]. Hence, a reliable calculation of the total neutron count rate requires an accurate knowledge of the amount of Cm isotopes in the fuel.

The objective of this work is to study the sensitivity of Cm abundances to uncertainties in the basic nuclear decay and cross section data, propagate these uncertainties through the burnup process and assess how they impact the uncertainties in Cm abundances in isotopic inventory problems.

II. DESCRIPTION OF THE ACTUAL WORK

1. Neutron Emission in Spent Fuel

The neutron emission of in spent fuel is originating from the decay of heavy elements. Heavy radionuclides in the fuel undergo α-decay and neutrons are produced by (α,n) reactions on oxygen present in oxide fuels. In addition, spontaneous fission in even-even isotopes is also a source of neutrons. For BU levels over 15 GWd/t and a cooling time over 2 years, the measured neutron count rate is mostly due to the spontaneous fission neutrons of ^{244}Cm [5].

The Cm isotopes are produced by successive neutron captures and β- decays from ^{239}Pu, which is formed from ^{238}U, by neutron capture and two successive β- decays. The main path to ^{242}Cm from ^{239}Pu starts with two successive neutron captures to ^{241}Pu, followed by a β- decay to ^{241}Am and a further neutron capture to ^{242}Am (or its metastable state, ^{242m}Am, which decays by isomeric transition) and then again a β- transition to ^{242}Cm. The main path to ^{244}Cm includes four successive captures from ^{239}Pu, to ^{243}Pu, followed by a β- decay to ^{243}Am and yet one more neutron capture and β- decay to ^{244}Cm [6]. In addition, ^{244}Cm can be produced by two successive neutron captures from ^{242}Cm and from ^{242}Am or ^{242m}Am if the neutron capture to ^{243}Am happens before the β- decay to ^{242}Cm. In such case, a further neutron capture and β-decay completes the path to ^{244}Cm. The different reaction paths are shown in Fig. 1.

As the build-up and decay path leading to Cm isotopes involves many transitions, there are many kinds of nuclear data whose uncertainties can impact the abundances of Cm in spent fuel. From these data we focused on decay data, half-lives and branching ratios, and radiative capture (n,γ) cross sections.

Uncertainties of these parameters are available in the nuclear data libraries. The nuclear data evaluation chosen as reference for this work is the ENDF/B-VII.1 release [7].

The goal of this work is also to assess the impact of the uncertainties on nuclear data on the production of Cm even isotopes. This is done by uncertainty propagation of nuclear data through the burnup process. For the uncertainty propagation in burnup processes we considered two methods. One method is based on first order perturbation theory approaches [8], not so expensive computationally but only valid for small uncertainties, since they involve linear approximation. The other method is based on a Monte Carlo approach [9] and requires more computation time. A first order perturbation theory approach was used for sensitivity and uncertainty studies on decay data, and a Monte Carlo approach was used for cross section data.

Prior to the uncertainty quantification, a sensitivity analysis was carried out. It helped to assess sensitivities of the system's responses to variations in input parameters. This is useful to gain insight about the data that drive the process and whose uncertainty can have a larger impact in the uncertainty of the Cm abundances.

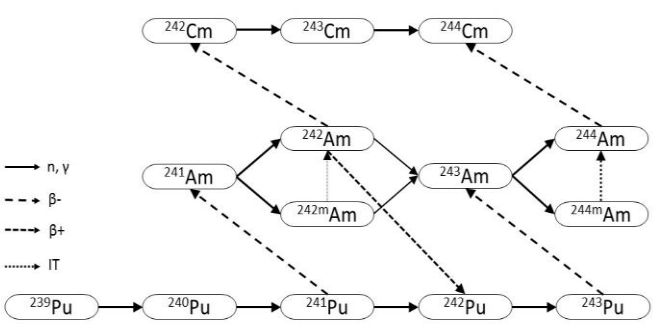

Fig. 1. Production paths leading to Cm isotopes.

2. Description of the model

The sensitivity studies and uncertainty propagation were conducted for a benchmark problem [10] using MOX fuel. MOX is interesting for this study because it has higher neutron emission than the conventional UO_2 due to the higher accumulation of neutron emitting isotopes [6]. The model is a simplified single pin, that keeps a fuel-to-moderator ratio of 0.5, as in a whole fuel assembly of a 17x17 PWR geometry. Light water with 600 ppm of boron was used as moderator. Zircaloy-2 was used as cladding, and its density was reduced to account for the air gap between fuel and cladding. The fuel was burned up to 48 GWd/t at constant power, in three irradiation cycles of 420 days, with intermediate decay periods of 30 days and a cooling time after discharge of 5 years. A summary of the composition of the fuel is displayed in Table I taken from [10]. The total ^{242}Cm and ^{244}Cm abundance for the reference case are 0.119 grams and 924.9 grams respectively.

Table I. Initial fuel composition

Isotope	w/o % in total U
^{234}U	0.001
^{235}U	0.250
^{238}U	99.749
Isotope	w/o % in total Pu
^{238}Pu	2.5
^{239}Pu	54.7
^{240}Pu	26.1
^{241}Pu	9.5
^{242}Pu	7.2
Total Pu/ U+Pu content	**8%**

III. SENSITIVITY AND UNCERTAINTY STUDIES ON DECAY DATA

1. Approach

Sensitivity analysis helps identifying the input parameters that are more important in a system. More particularly, sensitivity coefficients determine how the system's output parameters vary as a result of changes in the input parameters [11].

In this part of the study, sensitivity coefficients of the Cm abundances to half-lives and branching ratios of Pu, Am, and Cm isotopes, in the production paths to ^{242}Cm and ^{244}Cm, were calculated. The variation of the Cm abundance was assessed for each decay parameter independently. However, since the decay transitions studied have two decay modes (e.g. spontaneous fission and α-decay), their branching fractions are anti-correlated [12]. Therefore, we perturbed the branching fraction of the dominant process and adjusted the branching fraction of the secondary process accordingly. The sensitivity coefficients were then calculated as the change in Cm abundances resulting from the change in the ratio between the two branching fractions.

Sensitivity coefficients are equivalent to the derivative of the output parameter with respect to the input parameter at the reference value, but only under the assumption of a linear behaviour of the perturbation [11]. Fig. 2 shows changes in ^{242}Cm abundance due to changes in the ^{242m}Am half-life as an example of the linear behavior of the perturbations. For this purpose, finite perturbations were introduced in the decay parameters around their reference values. This was done using the COUPLE module in SCALE. Next, burnup simulations were carried out for every set of perturbed data with the TRITON module in SCALE [13].

Once the results of the Cm abundances for every set of perturbed data were obtained, we calculated the corresponding sensitivity coefficients.

We used Normalized Sensitivity Coefficients (NSC), as shown in (1) that compare the relative variation of the Cm abundance from its reference value to the relative variation of the half-life or branching fraction that was perturbed.

$$\mathrm{NSC} = \frac{\partial y / y}{\partial x / x} = \frac{(y_{pert} - y_{ref}) / y_{ref}}{(x_{pert} - x_{ref}) / x_{ref}} \qquad (1)$$

Besides the sensitivity analysis, sensitivity coefficients served to estimate the uncertainty in the Cm abundances due to the uncertainty in the decay data. Uncertainty propagation was carried out with a first order perturbation theory method, based on the use of sensitivity coefficients. Therefore, the use of this method requires that the output parameter changes linearly with the change of the perturbed input parameter [11]. This implies that its use is limited to nuclear data with small uncertainties [8].

The uncertainty quantification was carried out using the so-called Sandwich formula [14] that uses the sensitivity coefficients S and uses the variances of the input parameters as measure of their uncertainty, as shown in (2).

$$Var(y) = S \cdot Var(x) \cdot S^T \qquad (2)$$

Thus, the uncertainty σ(Cm) on the Cm abundances will be estimated from the uncertainty $\sigma_{T1/2}$ in the half-lives (or branching fractions) as in (3):

$$\sigma(Cm) = \sqrt{NSC \cdot (\sigma_{T1/2})^2 \cdot NSC} \qquad (3)$$

2. Results

The obtained results for the sensitivity coefficients and the uncertainty propagation of decay data are shown in Tables II and III. The sensitivity results show that the abundance of ^{242}Cm and ^{244}Cm are sensitive to their own half-lives. The abundances of both isotopes grow as their half-lives values are increased. However, ^{242}Cm abundance is more sensitive to its own half-life because ^{244}Cm undergoes α decay to ^{240}Pu, which will mostly build-up into ^{244}Cm again.

The results also indicate that the ^{242}Cm abundance is sensitive to the half-lives of ^{241}Pu and ^{242m}Am. ^{242}Cm builds up through the decay of ^{241}Pu to ^{241}Am, therefore if this process was slowed down, the rate of neutron capture would be increased for ^{241}Pu and a smaller share of atoms would decay to ^{241}Am. The same can be said about ^{242m}Am. Finally, results show that ^{242}Cm and ^{244}Cm are sensitive to the branching fraction of ^{242}Am, and their sensitivity coefficients have opposite sign. If ^{242}Am branching fraction to β- increases, so does the abundance of ^{242}Cm. On the other hand, if the branching fraction of β+ decay to ^{242}Pu increases, so does the abundance of ^{244}Cm.

Fig. 2 shows the changes in ^{242}Cm abundance due to changes in the ^{242m}Am half-life and the linear behaviour of these.

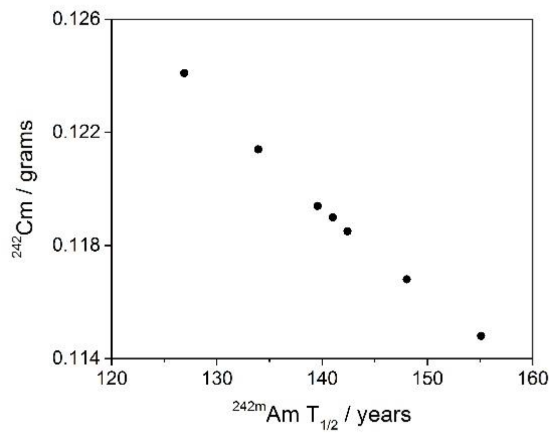

Fig. 2. ^{242}Cm abundance as function of the ^{242m}Am half-life.

The results of the uncertainty propagation of the considered decay data are shown in Table II and reveal that the associated uncertainties on Cm abundances are rather small. The combined uncertainty on the ^{242}Cm abundance is below 1%, while the one on the ^{244}Cm abundance is below 0.1%, as shown in Table V.

IV. SENSITIVITY AND UNCERTAINTY STUDIES ON CROSS SECTIONS

1. Approach

The cross section data that we considered were the radiative capture (n,γ) cross sections of Pu, Am, and Cm isotopes in the build-up chain leading to ^{242}Cm and ^{244}Cm.

The sensitivity studies for the cross sections were carried out in a similar fashion to the sensitivity studies for the decay data. We introduced finite perturbations in the nuclear data libraries using SANDY, a code developed at SCK•CEN [15]. Perturbations were introduced considering a 100% correlation between all the energy groups for each cross section, meaning that for every cross section perturbed, all the values for the different energy groups were perturbed to the same extent.

The burnup process was then simulated for every set of perturbed data and the Cm abundances were obtained. This was done by means of ALEPH, a code developed in SCK•CEN, which couples Monte Carlo particle transport with depletion calculations [16]. Results for the sensitivity studies are shown in Table IV. Fig. 3 shows the changes in the final Cm abundances due to perturbations in the ^{241}Pu cross section values.

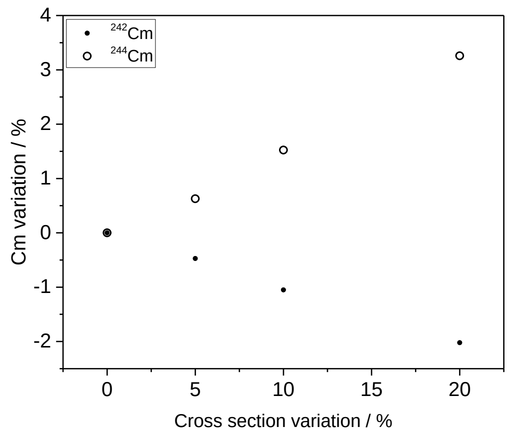

Fig. 3. Change in Cm abundances as a function of changes in ^{241}Pu (n,γ) cross section

After the sensitivity coefficients were calculated, we proceeded to the uncertainty propagation studies for the ones with the largest sensitivities.

The uncertainty propagation for cross section data was done using stochastic random sampling [8]. This method is not limited to small uncertainties because it does not involve any approximation. However, the statistical error has to be taken into account [8]. The first step consists in creating many sets of perturbed cross section data using their normal probability density functions. Then calculations for each of the sets of perturbed data are carried out and finally, output data are statistically analysed [11]. This is computationally expensive and thus, uncertainty propagation was limited to the cross sections to which Cm abundances yield a response that is not shadowed by the statistical error inherent to the method.

SANDY was used for the uncertainty propagation of the cross sections. SANDY makes use of the theory of stochastic sampling to propagate nuclear data covariance. It takes best estimates (File 3) and covariance matrices (File 33) from nuclear data evaluations and samples the data according to a normal probability density function, making sure they are distributed in the input phase space of the original covariance matrices. SANDY works with every nuclear code compatible with the ENDF-6 format [15]. For the uncertainty propagation 300 randomly sampled files were produced for every studied cross section. The burnup of the fuel was simulated with ALEPH.

Table II. Sensitivity of ^{242}Cm and ^{244}Cm abundances to half-lives and resulting uncertainties

Nuclide	$T_{1/2}$ (y)	Uncertainty (%)	NSC ^{242}Cm	NSC ^{244}Cm	σ(^{242}Cm) (%)	σ(^{244}Cm) (%)
^{241}Pu	14.29	0.04	-0.94	-0.01	0.04	<0.01
^{242m}Am	141.00	1.4	-0.39	<0.01	0.55	<0.01
^{242}Cm	0.446	0.04	5.73	<0.01	0.21	<0.01
^{244}Cm	18.10	0.17	<0.01	0.19	<0.01	0.03

Table III. Sensitivity of ^{242}Cm and ^{244}Cm abundances to branching fractions and resulting uncertainties

Nuclide	Branching fractions	Uncertainty (%)	NSC ^{242}Cm	NSC ^{244}Cm	σ(^{242}Cm) (%)	σ(^{244}Cm) (%)
^{242}Am	82.7 / 17.3	1.7	-0.17	<0.01	0.30	0.03

Table IV. Sensitivity coefficients of ^{242}Cm and ^{244}Cm abundances to radiative capture cross sections of different nuclides and resulting uncertainties

Nuclide	NSC ^{242}Cm	σ(^{242}Cm) (%)	NSC ^{244}Cm	σ(^{244}Cm) (%)
^{239}Pu	0.16	0.27	0.11	0.31
^{240}Pu	0.19	0.42	0.04	0.30
^{241}Pu	-0.11	0.50	0.15	0.64
^{242}Pu	0.01	0.01	0.48	9.6
^{241}Am	0.43	2.1	0.01	0.28
^{242m}Am	-0.05	1.2	0.01	0.27
^{243}Am	-0.01	<0.01	0.55	2.2
^{242}Cm	-0.01	0.10	0.01	<0.01
^{244}Cm	-0.01	0.01	-0.15	1.7

2. Results

The results of the simulations carried out with SANDY+ALEPH show that the Cm abundances are sensitive to the (n,γ) cross sections of several isotopes. Results for the sensitivity studies are shown in Table IV.

Results for the sensitivity studies show that ^{242}Cm abundance is most sensitive to ^{241}Am capture cross section. This cross section has a direct impact on the production of ^{242}Cm concentration because the radiative capture on ^{241}Am is the dominating process. Subsequently, in ^{242}Am the β-decay is largely dominating, transmuting into ^{242}Cm with a half-life of 0.6 days. The ^{242}Cm abundance is also sensitive to the capture cross sections of ^{239}Pu, ^{240}Pu and ^{241}Pu. Sensitivities to the to the capture cross sections of ^{239}Pu and ^{240}Pu are positive, meaning that an increase in the cross sections will induce an increase in the ^{241}Pu, which will decay to ^{241}Am and then continue the build-up chain to ^{242}Cm. The sensitivity to ^{241}Pu capture cross section is negative, meaning that the more neutron capture in ^{241}Pu the less ^{241}Am is formed and therefore less ^{242}Cm.

The data in Table IV also indicate that the abundance of ^{242}Cm is little sensitive to its own radiative capture cross section. This behaviour is due to the fact that the production of ^{242}Cm is mainly driven by other processes such as the decay of ^{242m}Am and ^{242}Cm, as explained earlier.

Results also show that ^{244}Cm is sensitive to ^{242}Pu and ^{243}Am cross sections. Both nuclides have very small decay rates compared to their neutron capture rates. This means that the main way of disappearing from the system is by neutron capture. Besides, ^{243}Pu and ^{244}Am resulting from these captures undergo β- decay almost immediately, resulting in a direct impact of the cross sections in ^{244}Cm concentration.

Additionally, sensitivity results show that ^{244}Cm is also sensitive to ^{239}Pu, ^{240}Pu and ^{241}Pu cross sections. Unlike the case of ^{242}Cm, now the three sensitivity coefficients are positive, meaning that an increase in the cross section for ^{241}Pu will increase the production of ^{242}Pu that leads to the build-up of ^{244}Cm.

The data in Table IV also indicate also that ^{244}Cm is sensitive to its own radiative capture cross section. We can conclude that the radiative capture process on ^{244}Cm plays a role in the build-up of ^{244}Cm similar to the role played by its decay.

The relative uncertainty results are also shown in Table IV. The contributions of ^{241}Am and ^{242m}Am cross sections to ^{242}Cm uncertainty are the most important. The contribution due to the uncertainty on ^{241}Am radiative capture cross section is 2.1% and is due to the sensitivity of ^{242}Cm abundance to ^{241}Am cross section, with a sensitivity coefficient of 0.43. The contribution due to the uncertainty on ^{242m}Am radiative capture cross section is 1.2% and it results mainly from the uncertainty of its cross section. Fig. 4 shows the relative uncertainty of the radiative capture cross section of some isotopes for different ranges of incident neutron energies. The neutron flux for the whole energy spectrum in logarithmically spaced energy bins is shown in Fig. 5.

The contributions of ^{242}Pu and ^{243}Am cross sections to ^{244}Cm uncertainty are the most important. Uncertainty of 2.2% of ^{243}Am can be explained due to its large sensitivity coefficient. The contribution of ^{242}Pu is much higher (9.6%). This results both from the high sensitivity in ^{244}Cm production to ^{242}Pu (n,γ) cross section and uncertainty of the ^{242}Pu (n,γ) cross section. The uncertainty of the ^{242}Pu (n,γ) cross section is larger than the one for ^{243}Am as shown in Fig. 4. The impact of the radiative capture cross section uncertainty on ^{244}Cm is also non negligible (1.7%).

Finally, we estimated the overall uncertainty on ^{242}Cm and ^{244}Cm due to the considered decay and cross section data and their uncertainties and the results are given in Table V. The given uncertainties were calculated as the square root of the sum of the individual variances. The overall impact of the uncertainty on radiative capture cross sections data on ^{242}Cm is about 2.5% and on ^{244}Cm is about 10%.

Table V. Total uncertainty

Uncertainty source	^{242}Cm uncertainty (%)	^{244}Cm uncertainty (%)
Decay data	0.7	<0.1
Cross sections	2.5	10

V. CONCLUSIONS

The objective of the work was to assess the sensitivity of Cm abundances to uncertainties in the basic nuclear decay and neutron cross section data and, more particularly, assess how these uncertainties affect the abundances of ^{242}Cm and ^{244}Cm, two of the main contributors to neutron emission in the spent fuel. Sensitivity analysis and uncertainty propagation were carried out for a benchmark MOX fuel case. SCALE was used for the sensitivity analysis of the decay data. SANDY was used for the uncertainty propagation of the cross section data, using ALEPH for the transport and depletion simulations.

Results show that the uncertainties in the decay data do not have a significant contribution when compared to the uncertainties in the cross section data. The contribution of the uncertainty of ^{242m}Am half-life is about 0.6% and is the largest from all the decay data. A better understanding and possible reduction of the uncertainty on this parameter would be beneficial.

The combined uncertainty due to the considered radiative capture cross section on ^{242}Cm is about 2.5%. Uncertainties in the cross sections of ^{242m}Am and ^{241}Am are the largest. In the case of ^{242m}Am, this impact is mainly due to its relatively large uncertainty in the cross section data rather than due to a large sensitivity. It is expected, therefore, that an improvement in the ^{242m}Am capture cross section results in a reduction of the uncertainty component on the abundance of ^{242}Cm.

The combined uncertainty due to the considered radiative capture cross section on ^{244}Cm is about 10% Uncertainties in radiative capture cross sections of ^{242}Pu, ^{243}Am and ^{244}Cm are the main contributors to the uncertainty in ^{244}Cm abundance. We could determine that the uncertainty on ^{242}Pu radiative capture cross sections has the largest impact. It is expected, therefore, that an improvement in the ^{242}Pu capture cross section results in a reduction of the uncertainty component on the abundance of ^{244}Cm.

The obtained results allow to determine the uncertainty due to nuclear data on Cm isotopes and, since neutron emission of a SFE is usually dominated by ^{242}Cm and ^{244}Cm, on the predicted response of neutron detectors used in spent fuel measurements. Future work will focus on sensitivity and uncertainty studies on UO_2 fuel. In addition the impact of irradiation history parameters such as the burnup will be studied.

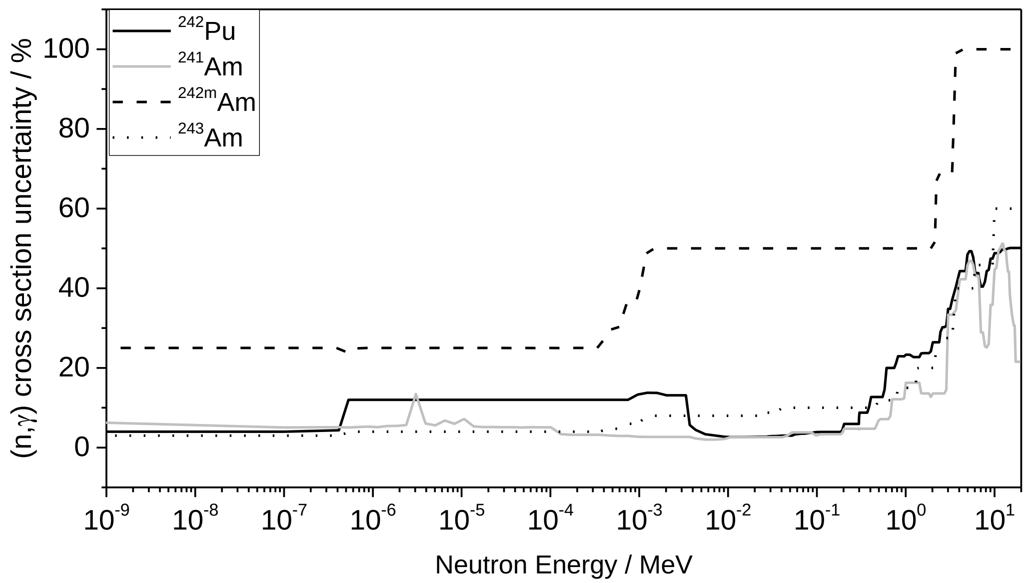

Fig. 4. Relative uncertainties of some of the considered radiative capture cross sections.

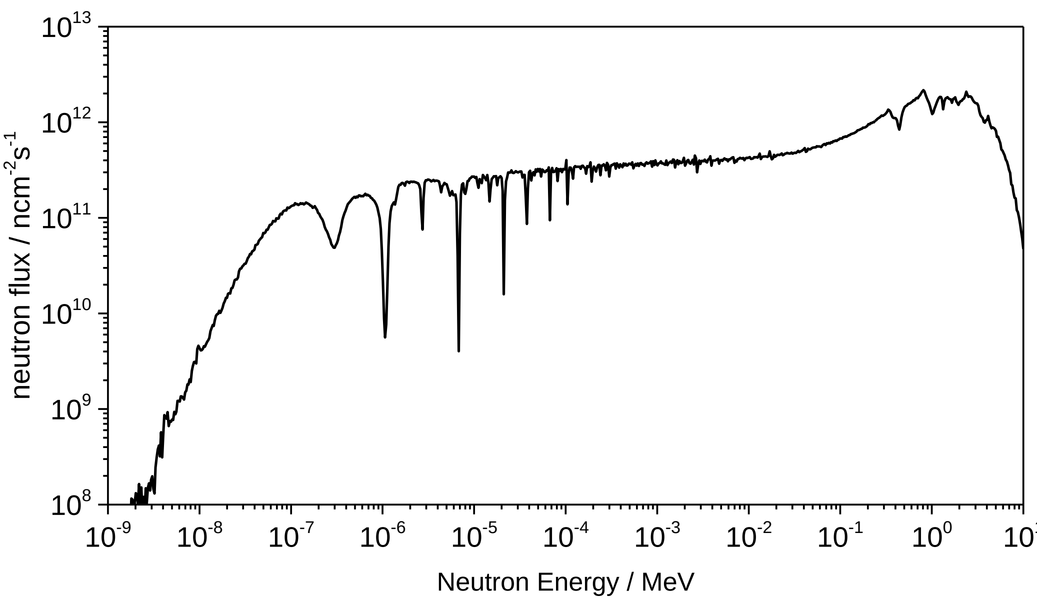

Fig. 5. Neutron flux spectrum in the studied model.

.

REFERENCES

1. IAEA, “Safeguards Techniques and Equipment: 2011 Edition,” International Nuclear Verification Series. No. 1. Rev. 2 (2011).
2. S. HSUE, T. CRANE, W. TALBERT, J. LEE, “Nondestructive Assay Methods for Irradiated Nuclear Fuels”. Los Alamos National Laboratory of the University of California (1978).
3. A. BORELLA. “The Fork Detector for Spent Fuel Measurements: Measurements and Simulations”. *Proceeding of the INMM Annual Meeting* (on the website: http://www.inmm.org//AM/Template.cfm?Section=Meeting_Home), Ann Arbor, Michigan, United States, 15-19 July 2012 / Institute of Nuclear Materials Management, Chicago, IL, United States, INMM (2012), p. 1-9.
4. S. VACCARO et al. “A new Approach to Fork Measurements Data Analysis by RADAR-CRISP and ORIGEN integration”, *IEEE Transactions on Nuclear Science*, (2014).
5. D. REILLY, N. ENSSLIN, H. SMITH, S. KREINER, *Passive Nondestructive Assay of Nuclear Materials*, p.340, U.S. Nuclear Regulatory Commission, Washington D.C. (1991).
6. A. SASAHARA, T. MATSUMURA, G. NICOLAOU, D. PAPAIOANOU. “Neutron and Gamma Ray Source Evaluation of LWR High Burn-up UO2 and MOX Spent Fuels”. *Journal of NUCLEAR SCIENCE and TECHNOLOGY*, **Vol 41**, *No. 4*, p. 448 (2004).
7. M.B. Chadwick et al. “ENDF/B-VII.1 nuclear data for science and technology: cross sections, covariances, fission product yields and decay data”. *Nuclear Data* Sheets, **Vol 112.12** Special Issue on ENDF/B-VII.1 Library, p.2887 (2011).
8. C.J. DÍEZ, O. BUSS, A. HOEFER, D. PORSCH, O. CABELLOS, “Comparison of nuclear data uncertainty propagation methodologies for PWR burn-up simulations”, *Annals of Nuclear Energy*, **Vol.77**, p. 101 (2015).
9. N. GARCÍA-HERRANZ, O. CABELLOS, J. SANZ, J. JUAN, J. C. KUIJPER, “Propagation of statistical and nuclear data uncertainties in Monte Carlo burn-up calculations”, *Annals of Nuclear Energy*, **Vol. 35**, p. 714 (2008)
10. O'CONNOR, G.J. “Burn-up credit criticality benchmark Phase 4-B: results and analysis of MOX fuel depletion calculations”. Nuclear Energy Agency of the OECD (NEA): Organisation for Economic Co-Operation and Development - Nuclear Energy Agency (2003).
11. N. GARCÍA-HERRANZ. “Nuclear Data Sensitivity/Uncertainty (S/U) analysis in nuclear applications”. Contribution at GENTLE Nuclear data processing and use in nuclear applications Course, Geel, Belgium, 14-18 November (2016).
12. J. MARTÍNEZ et al., “Propagation of Neutron Cross Section, Fission Yield, and Decay Data Uncertainties in Depletion Calculations”. *Nuclear Data Sheets*, **Vol 118**, p. 480 (2014).
13. *SCALE: A Comprehensive Modelling and Simulation Suite for Nuclear Safety Analysis and Design*, ORNL/TM-2005/39, Version 6.1, Oak Ridge National Laboratory, Oak Ridge, Tennessee, June 2011.
14. D. CACUCI, *Sensitivity and Uncertainty Analysis*, p.122, Ed Chapman & Hall, Boca Raton, Florida (2003)
15. L. FIORITO, G. ŽEROVNIK, A. STANKOVSKIY, G. VAN DEN EYNDE, P.E. LABEAU, “Nuclear data uncertainty propagation to integral responses using SANDY”, *Annals of Nuclear Energy*, Vol.101, p. 359 (2017).
16. VAN DEN EYNDE G., STANKOVSKIY A., MALAMBU MBALA E., VIDMAR T. “ALEPH2 Monte Carlo burn up code”. *Proceedings of the International Conference on Nuclear Criticality Safety*, Edinburgh, United Kingdom, 19-23 September 2011, Paris, France, OECD/NEA, (2011).